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**Detroit Edison**
*A DTE Energy Company*

10CFR50.73

January 31, 2002
NRC-02-0008

U S Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Licensee Event Report (LER) No. 01-004

Pursuant to 10 CFR 50.73(a)(2)(iv)(A), Detroit Edison is submitting the enclosed LER No. 01-004. This LER documents a manual scram and the automatic actuation of other systems listed in paragraph (a)(2)(iv)(B).

No commitments are being made in this LER.

Should you have any questions or require additional information, please contact Mr. Norman K. Peterson of my staff at (734) 586-4258.

Sincerely,

William T. O'Connor Jr.

cc: T. J. Kim
M. A. Ring
M. V. Yudasz, Jr.
NRC Resident Office
Region III
Regional Administrator, Region III
Wayne County Emergency Management Division

IE22

| NRC FORM 366 (7-2001) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004 |
|---|---|---|
| **LICENSEE EVENT REPORT (LER)** (See reverse for required number of digits/characters for each block) | | Estimated burden per response to comply with this mandatory information collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection. |

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**4. TITLE**
Manual Reactor Scram Due to Loss of Stator Water Cooling

| 5. EVENT DATE | | | 6. LER NUMBER | | | 7. REPORT DATE | | | 8. OTHER FACILITIES INVOLVED | |
|---|---|---|---|---|---|---|---|---|---|---|
| MO | DAY | YEAR | YEAR | SEQUENTIAL NUMBER | REV NO | MO | DAY | YEAR | FACILITY NAME | DOCKET NUMBER 05000 |
| 12 | 06 | 2001 | 2001 | — 004 | — 00 | 01 | 31 | 02 | FACILITY NAME | DOCKET NUMBER 05000 |

| 9. OPERATING MODE | 1 | 11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | 20.2201(b) | | 20.2203(a)(3)(ii) | | 50.73(a)(2)(ii)(B) | | 50.73(a)(2)(ix)(A) |
| 10. POWER LEVEL | 100 | | 20.2201(d) | | 20.2203(a)(4) | | 50.73(a)(2)(iii) | | 50.73(a)(2)(x) |
| | | | 20.2203(a)(1) | | 50.36(c)(1)(i)(A) | X | 50.73(a)(2)(iv)(A) | | 73.71(a)(4) |
| | | | 20.2203(a)(2)(i) | | 50.36(c)(1)(ii)(A) | | 50.73(a)(2)(v)(A) | | 73.71(a)(5) |
| | | | 20.2203(a)(2)(ii) | | 50.36(c)(2) | | 50.73(a)(2)(v)(B) | | OTHER Specify in Abstract below or in NRC Form 366A |
| | | | 20.2203(a)(2)(iii) | | 50.46(a)(3)(ii) | | 50.73(a)(2)(v)(C) | | |
| | | | 20.2203(a)(2)(iv) | | 50.73(a)(2)(i)(A) | | 50.73(a)(2)(v)(D) | | |
| | | | 20.2203(a)(2)(v) | | 50.73(a)(2)(i)(B) | | 50.73(a)(2)(vii) | | |
| | | | 20.2203(a)(2)(vi) | | 50.73(a)(2)(i)(C) | | 50.73(a)(2)(viii)(A) | | |
| | | | 20.2203(a)(3)(i) | | 50.73(a)(2)(ii)(A) | | 50.73(a)(2)(viii)(B) | | |

**12. LICENSEE CONTACT FOR THIS LER**

| NAME | TELEPHONE NUMBER (Include Area Code) |
|---|---|
| Jerome Flint – Principal Specialist, Licensing | 734-586-5212 |

**13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT**

| CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX | CAUSE | SYSTEM | COMPONENT | MANU-FACTURER | REPORTABLE TO EPIX |
|---|---|---|---|---|---|---|---|---|---|
| E | TJ | HX | E275 | Y | | | | | |

| 14. SUPPLEMENTAL REPORT EXPECTED | | | | 15. EXPECTED SUBMISSION DATE | MONTH | DAY | YEAR |
|---|---|---|---|---|---|---|---|
| | YES (If yes, complete EXPECTED SUBMISSION DATE). | x | NO | | | | |

**16. ABSTRACT** (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On December 6, 2001, at 0350 hours operators inserted a manual scram from 100 percent power due to a leak on the Stator Water Cooling system upper heat exchanger. The leak occurred when a vent line separated from the heat exchanger during removal of the pipe cap from the vent line. All systems responded as expected and all control rods fully inserted. Reactor level was recovered with the Feedwater/Condensate System. No Emergency Core Cooling Systems initiated and no Safety Relief Valves lifted. The cause of the vent line failure was inadequate problem documentation and resolution that dates back to 1996. The original vent line design was a contributing factor. This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in manual or automatic actuation of any systems listed in paragraph (a)(2)(iv)(B), (i.e., actuation of the Reactor Protection System including reactor scram or reactor trip).

The original vendor supplied heat exchanger vent lines were strengthened and an unsupported length was reduced. Schedule 80 steel was used to replace the schedule 40 steel and the connection to the heat exchangers was welded vice threaded.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

## Initial Plant Conditions:

| | |
|---|---|
| Mode | 1 (Power Operation) |
| Reactor Power | 100 Percent |
| Reactor Pressure | 1025 psig |
| Reactor Temperature | 540 Degrees Fahrenheit |

## Description of the Event

On December 6, 2001 a Nuclear Operator [utility non-licensed] attempted to draw a water sample from the standby stator water cooling (SWC) heat exchanger vent line to verify conductivity after flushing the system during the Eighth Refueling Outage. When removing the pipe cap (PSF) the vent line broke at the threaded connection to the heat exchanger.

The SWC system pressure dropped from approximately 62 to 40 psig and control panel annunciators alerted Control Room Operators [utility licensed]. System inventory decreased through the broken vent line. The standby line up for the heat exchanger requires the discharge valve be closed and inlet valve open, maintaining the heat exchanger pressurized to system pressure.

In anticipation of an automatic reactor scram due to a turbine trip from loss of SWC, a manual reactor scram was initiated by placing the mode switch in the shutdown position at 0350 hours. All safety systems responded as expected. All rods fully inserted into the core. Reactor water level decreased below Level 3 (approximately 173 inches above the top of active fuel) as expected, resulting in Primary Containment Isolation (JM) of Group 13, Drywell Sumps (WK). Primary Containment Isolation Groups 4, Residual Heat Removal Shutdown Cooling and Head Spray (BO), and 15, Traversing Incore Probe System (IG), received isolation signals, but were already isolated per normal lineup. Reactor level was recovered with the Feedwater/Condensate System (SG, SJ). No Emergency Core Cooling Systems initiated and no Safety Relief Valves lifted.

This event is being reported in accordance with 10 CFR 50.73(a)(2)(iv)(A), as an event that resulted in manual or automatic actuation of any systems listed in paragraph (a)(2)(iv)(B), (i.e., Reactor Protection System (RPS) including reactor scram or reactor trip).

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

**Cause of the Event**

The cause of the scram was a loss of SWC system inventory due to piping separation of the upper heat exchanger vent line. The Stator Water Cooling system circulates water through the main generator stator and stator terminals to remove the heat developed at each component.

The cause of the vent line failure was inadequate problem documentation and resolution. In April 1996, a work request identified a leak on the upper SWC heat exchanger where the vent line threaded into the top of the heat exchanger. The work request also requested that a means to restrain the piping be investigated. The vent line was tightened and the work request was closed with no additional restraints being required.

The SWC Heat Exchanger vent line was identified as being loose again in May 2000. Repairs were completed, including in-shop repairs to the vent line threads. The vent line was reinstalled with an additional 180 degrees of rotation from its first orientation, providing additional thread engagement, but possibly damaging the threaded area of the vent piping.

During Eighth Refueling Outage in November 2001, the main generator stator underwent copper oxide removal. A concern existed that the upper SWC Heat Exchanger may not have had all of the chemicals removed due to the normally closed discharge valve and lack of specific documentation showing the upper heat exchanger had been flushed. If chemicals were present in the heat exchanger and it was placed in service a potential existed for a generator ground. Based on the information provided to Operations, it was decided a sample should be taken from the upper SWC heat exchanger.

In order to draw a sample of water from the upper SWC Heat Exchanger a pipe cap needed to be removed. During removal the vent pipe at the threaded connection to the heat exchanger failed, resulting in loss of inventory from the SWC system.

The original vent line design was a contributing factor in the vent line failure. The vent line's configuration and material strength did not support normal use and service. Records indicated the vent was original vendor supplied equipment using schedule 40 piping which satisfied the code requirements and was adequate for the static loading of the line's weight. The SWC heat exchanger vent lines piping arrangement consisted of threaded ¾ inch schedule 40 stainless steel pipe, stainless steel elbow, a straight schedule 80 pipe, a valve, another straight schedule 80 pipe and an end cap (entire length about 20 inches). The upper SWC heat exchanger vent line valve weighed approximately 13 pounds (including flanges) with a 10-inch length of pipe between the elbow and valve.

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**Analysis of the Event**

This report is required by 10 CFR 50.73(a)(2)(iv)(A) because of the unplanned actuation of reportable systems. Specifically, the reactor protection system was manually actuated in anticipation of a turbine trip due to loss of the stator water coolant system.

A scram from turbine trip is an analyzed transient for which the plant is designed. All systems responded as expected. All rods fully inserted into the core. Reactor water level was maintained well above the top of active fuel by the Condensate/Feedwater Systems and pressure was maintained below design by the Turbine BypassValves. Reactor water level decreased below Level 3 (approximately 173 inches above the top of active fuel) as expected, but above level 2 (approximately 110 inches above the top of active fuel) resulting in Primary Containment Isolation of Group 13, Drywell Sumps. Primary Containment Isolation Groups 4, Residual Heat Removal Shutdown Cooling and Head Spray, and 15, Traversing Incore Probe System, received isolation signals, but were already isolated per normal lineup. No Emergency Core Cooling Systems initiated and no Safety Relief Valves lifted.

This event did not affect the ability of systems required to maintain safe shutdown conditions, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident. Based on the preceding, it is concluded that there was no adverse impact on safety as a result of this event.

**Corrective Actions**

The SWC heat exchanger vent lines were strengthened and an unsupported length was reduced. Schedule 80 steel was used to replace the schedule 40 steel and the connection to the heat exchangers was welded vice threaded. Use of schedule 80 steel is a typical practice at Detroit Edison.

A historical search for vent and drain line failures was performed. No instances of vent or drain line failures as a result of pipe cap installation or removal were found. Several instances of vibration induced failures were found, but vibration is not suspected to have played a role in this failure.

Further corrective actions relating to this event are being considered, and will be developed and implemented commensurate with established priorities and processes of the Fermi 2 corrective action program. This event is documented in the Fermi 2 corrective action program as CARD 01-22371. One such corrective action is review for similar configurations in other systems, focusing on vendor supplied systems.

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**Additional Information**

A. Failed Components

Component: Vent Line on Heat Exchanger N3033-B027
Description: Upper Stator Cooling Water Heat Exchanger Vent Line
Manufacturer: General Electric Company (GEC) Turbine-Generator, Ltd. of Rugby, England
Type: NA

B. Previous LERs On Similar Problems

No similar failures were found.